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An Assessment of the Penetrations in the First Wall Required for Plasma Measurments for Control of an Advanced Tokamak Plasma Demo

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AN ASSESSMENT OF THE PENETRATIONS IN THE FIRST WALL REQUIRED FOR PLASMA MEASUREMENTS FOR CONTROL OF AN ADVANCED TOKAMAK PLASMA DEMO.

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Abstract: A Demonstration tokamak (Demo) is an essential next step toward a magnetic-fusion based reactor. One based on advanced-tokamak (AT) plasmas is especially appealing because of its relative compactness. However, it will require many plasma measurements to provide the necessary signals to feed to ancillary systems to protect the device and control the plasma. This note addresses the question of how much intrusion into the blanket system will be required to allow the measurements needed to provide the information required for plasma control. All diagnostics will require, at least, the same shielding designs as planned for ITER, while having the capability to maintain their calibration through very long pulses. Much work is required to define better the measurement needs and the quantity and quality of the measurements that will have to be made, and how they can be integrated into the other tokamak structures.

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I. INTRODUCTION

There is a trend in fusion physics to try to move to smaller devices to achieve a cheaper device for the ultimate reactor. The exemplar of this trend is the so-called “Advanced Tokamak” (AT) where the physics results might result in a smaller plasma with improved performance. However, in existing and future devices, this kind of approach requires very sophisticated control schemes based on the measurement of many plasma parameters to feed information to actuators of various fuelling, heating and current-drive sources. Additional magnetic field coils, inside the vacuum vessel to have reasonable response times, will also almost certainly be necessary[1]. In today’s experiments, a variety of so-called AT scenarios have been developed, and overall there is a general trade-off between possible compactness and complexity, i.e. one can devise a very compact system, but at the expense of having a very complex and fully integrated scheme.

For a Demo device or a reactor, it may not be possible to decrease the dimensions of the containment vessel as much as a decrease in the plasma size. Apart from such impacts as the first-wall heat loads and neutron radiation damage on materials, the blankets require significant volume to breed the necessary tritium and useful heat. There will also have to be penetrations through the blankets to permit the presence of sufficient diagnostics and other ancillary equipment. The present design concept of an advanced tokamak version of a Demo in the United States[2] allocates 2.5 m^2 of the first wall for control of the plasma, presumably for measurement and responding actuator systems. This corresponds to only about 1% of the outer blanket area, but most of the required space is close to the horizontal mid-plane. The allocation compares with ~3.3 m^2 for a single equatorial port on ITER. In Japan, for the SlimCS design, first-wall space of ~ 1.5 m^2 is allocated for "diagnostics" with 6 m^2 allocated to the heating systems[3]. Note that the engineering for the penetrations for Demo, where most of the blanket shielding is needed for breeding tritium, will be very much more complex than for ITER which has shielding blankets which are mostly passive.

The main goal of this note is to begin a process whereby the design of control systems for the Demo will be integrated from the start into the mechanical design of the tokamak. It is being written because of concern that much too little space is presently considered adequate, for instance by the ARIES-AT designers[2], and that a much larger requirement will cause a severe conflict with the needs of tritium breeding. Since the impact on the overall tokamak device design is likely to be very significant, it is important to make an initial attempt to scope the problem, while being very aware that the final set of diagnostic measurements, with the necessary specification on their resolutions, cannot be fully established until well into the ITER operational period. Hence a set of measurements which, in the author's view, will be needed for the control of the plasma, have been identified. So have the diagnostics which, from present experience, are most likely to be used. To help quantify the overall space envelope the actual open apertures presently planned for diagnostics on ITER have been used, but a factor has been applied to account for the additional material thicknesses surrounding the apertures, including labyrinths, shutters, calibration devices and their controllers, which is space taken out of the somewhat complex active blanket systems. Hopefully the numbers arrived at will give good initial guidance which will obviously be improved over time.

There are many other feasibility issues for diagnostic components such as enabling magnetic diagnostics near the plasma to perform in the high radiation and thermal environment, survival of mirrors mounted close to the first wall, and for operation with good calibration for the extremely long pulse-lengths. These are critical issues but they will not be addressed here. It should also be noted that installing diagnostics as close as possible to the first wall (e.g. fast magnetic coils) with some shielding to provide operational capability and survival, will also intrude into the blankets.

Very good information on the plasma profiles inside the core plasma will probably be required in supporting the control of an AT plasma. (One expects that the operational scenario to be used will have been determined during operation of ITER burning plasmas.) This need brings into question how well these measurements can be made with the radiation environment and the inability of a low-energy neutral beam or of pellets to penetrate deeply. Obviously one can only speculate now on the resolution and accuracy required of the measurements, but it is clear that active programs to determine the definitive requirements for the measurements and relevant diagnostics and their penetrations, on ITER and other devices, will be necessary. Modeling using synthetic diagnostics could also significantly help to improve the measurement definition.

In summary, a push towards compactness has many consequences, which have to be integrated and/or mitigated:

- Increased control needs (complexity),
- reduced access for diagnostics and ancillary systems,
- reduced operating margins,
- increased inability to use some established diagnostic techniques (e.g. beam assisted).

So far, no consistency study has been performed in any AT Demo design studies in how far compactness can be pursued, and how the plasmas can be controlled.

II. NECESSARY ACCESS SPACE FOR DIAGNOSTICS

All diagnostics have to be integrated onto the tokamak in such a way as to minimize nuclear radiation levels at the magnetic field coils, to protect them, and to minimize the activation levels to allow closer approach to the device and easier maintenance outside it. Hence penetrations through the structure are labyrinthine and as small as possible consistent with the measurement requirement. Optical systems use reflective and possibly diffractive optics, rather than refractive. Electrical systems must be designed with carefully selected ceramic insulation with sufficient thickness to limit the impact of transient radiation-induced effects. Such thinking has been incorporated into conceptual designs for diagnostics for ITER[4], so these designs can be used as a basis for assessing the requirements for the equivalent diagnostic system for Demo. These conceptual designs have not yet included some essential elements such as shutters or calibration devices, so that, if anything, the size defined for ITER by the clear aperture is smaller than diagnostics' intrusions into the passive blanket.

Another major contributor to the total spatial occupation of diagnostics within the shielding wall is the selection of the diagnostic techniques, and the spatial resolutions and coverage of the plasma that are necessary for the control of the plasma. The trend on currently operating devices is to add more systems with better spatial resolution, but this trend must be reversed for the Demo device. Since the operational plasma properties of ITER and an AT Demo tokamak are quite similar, detailed studies can be done in the later operational period of ITER to define exactly the operational range for Demo and the quality of instrumentation sufficient for its control. The neutron flux and fluence at the first wall will be much higher in Demo (flux 5 – 10 times larger).

The third contributor is the presence of more than one diagnostic apparently measuring the same physics parameter. The multiplicity might be due to different time behavior of the measurement systems (e.g. pulsed versus steady), different physical processes on which the measurements are based (one might be very dependent on relativistic effects), or inability to function satisfactorily over the full operational range of the device[5]. This multiplicity must be reduced for Demo, but it will certainly have to be considered in ensuring coverage of the full operational range of the device.

For the purpose of this paper, a set of diagnostics has been chosen as the necessary control set. With time and operational experience, the selection may change but it is unlikely that the total space requirement can be reduced further. A suggested set of the measurements of plasma parameters thought to be necessary for Demo operation is shown in Table 1. The set has been divided into two groups, those needed for device protection and those needed for plasma control though the boundary between the groups is not very firm. It has been assumed that the device will move forward to full operation with D-T fuel quickly and no extra instrumentation has been considered for a period of physics study to establish operational regimes, ITER having hopefully provided the necessary experience.

In considering measurements in Demo, it must be realized that the device will have to operate very close to its operational limit. The plasma pressure will be close to the beta-limit and the divertor will mostly operate in the detached-mode. Experience suggests that it will not be possible to operate without some occasional “transient events”, but if one occurs the plasma will have to recover and the pulse continue. Some of the measurements listed in table 1 reflect this thinking.

The specific diagnostics that could be chosen to provide the listed minimum sets of measurements are shown in Tables 2a and 2b. The intrusion area through the blankets required by each diagnostic (i.e. that area not available for tritium breeding) has been estimated very crudely on the basis of the size currently allocated in the ITER design concepts[6]. In some cases, reduced spatial resolution with respect to ITER has been assumed. In other cases no penetrations are thought to be necessary, but their design will have to be considered during the tokamak design. The wiring for magnetic diagnostics can probably enter the vessel outside the blanket region and vacuum diagnostics will be housed on vacuum ducts, again away from the blanket region. (Note that it should not be assumed that magnetic diagnostics will not require some volume from the blankets so that they can be mounted with some protection and sufficient frequency response.) Some measurements, though critical for physics now, cannot depend on the same techniques: the inability of a low energy (~100 keV/amu) neutral beam to penetrate the plasma for diagnostics limits the ability to measure the ion temperature and alpha-ash. This paper makes use of the ARIES-AT assumption that no neutral beam heating or current drive will be applied[7]. This lack would negate the use of motional Stark effect (MSE) for measurement of the current density distribution. The use of advanced analysis codes and the development of the use of synthetic diagnostics may reduce the total measurement requirement assumed now.

The sizes of intrusions into the blankets for the penetrations for individual diagnostics have mostly been scaled up from sketches of the ITER equivalent by a factor of two to allow space for wall material, for structures and labyrinths, and to allow space for calibration and shutter devices within the complex blanket configurations. Diagnostics sharing the same port structure would certainly use such space. In the case of some measurements, there is no currently known technique for a Demo: an area of 0.1 m^2 has been assigned arbitrarily for a potential solution.

Hence total penetration areas of 1.15 m^2 for device protection and a further area of 1.72 m^2 are required for this minimal measurement capability; a total penetration area of 2.9 m^2. Some of the assumptions used to reach this number can be challenged, but it is more likely that the control needs will require greater measurement capability, and that the scaling factor used for the apertures is generally too small. Hence a guidance requirement of at least 3 m^2 for diagnostics should be used in the initial concept design of a real Demo device. One should also note that all these diagnostics cannot all be assigned to one port location because of the different functions that they serve. Some may be installed on top or bottom ports of the device, clear of the blanket volume, but that will have to be worked into the design of the tokamak. Such access has already been considered for the vacuum pumping system in ARIES-AT[7].

A much more detailed assessment of both the physics/control selection of diagnostic measurements and the requirements of each diagnostic, including engineering design of the blanket/penetration interfaces, is clearly urgently required to allow preliminary design of the tokamak itself. But the final definition of the diagnostics, taking account of new physics understanding of the plasma and better understanding of the operation in the Demo environment will take a long time..

As an aside, a similar assessment should be made for the heating and current drive systems, the actuator systems responding to the diagnostic information. These systems play many roles in start-up, heating, establishing the operational mode, current drive and instability mode-stabilization, so it is likely that more than one technique will be applied. The authors of the physics part of the ARIES-AT study chose ICRF fast wave for central current drive and LH for off-axis current drive, without additional heating to achieve the H-mode[8]. The Lower Hybrid system was assumed to provide 37 MW of power in a first-wall area of 1.26 m^2 with some additional 5 MW of High Harmonic Frequency heating. These numbers appear to be very optimistic, and do not include any Electron Cyclotron component. Hence a similar evaluation of the space requirements for all the actuator responders, including fueling systems, to the control signals from the diagnostics is highly desirable. The space allocated for heating for the Japanese SlimCS device[3], assumed to be by neutral beams, is 6 m^2.

III. CONCLUSIONS

This note is intended to highlight the potential conflict between the needs of plasma control and of tritium breeding in the first wall of the Demo device. It suggests that current designs, e.g. ARIES-AT, do not provide sufficient access for the necessary plasma measurement systems, though it is clear that there is little freedom to provide more because of the requirements for tritium breeding.

The estimates made for the penetration needs for plasma diagnostics must be treated as the first step in trying to define their needs at the first wall. Despite the rather naïve assumptions for the required penetration areas for measurements to ensure protection of the device and to enable control of the plasma, it is clear that allocating space for them must be a priority of the device design. It is suggested that a total area of at least 3 m^2, distributed to the locations of the first wall required to fulfill the function of the individual diagnostics, must be allocated for measurement. A similar assessment of the area needed to provide the responding heating and fuel injection systems should also be made, and this will clearly add to the volume made unavailable to breeding of tritium

It is clearly essential that the requirements for the control of the tokamak plasma be taken into account properly in the design of Demo. Hence much more detailed effort is required to define exactly what those requirements will be and to establish what access will really be necessary. Operational physicists on tokamaks, including ITER, must close in on the appropriate operational scenario for Demo, and then on the minimum quantity and quality of the measurements needed for the control of the plasma. On the part of diagnostic developers, consistency between the measurement requirements and the diagnostic capability and design must be established. They must also develop new techniques to compensate for those which will be incompatible with the Demo environment. Tokamak designers must include all these requirements at the earliest possible stages of device design.

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TABLES

Table 1: Proposed measurements for machine protection and control.

Machine Protection	**Contributing plasma measurement**
Disruptive instabilities	Disruption precursors, large ELMs
Intense local heat load	High local first wall/divertor plate temperatures
Plasma beta	Magnetic or sum of kinetic measurements
Start-up initial conditions	Pressure and partial pressures
Start-up localization	Magnetic equilibria
Profile control (auxiliary heating, fuelling)	Density, temperature, rotation profiles
Development of excess alpha-particle loss	Lost- alphas, high local wall temperature
Significant loss of plasma (disruption, etc.)	Many of the above measurements
Plasma Control	
Plasma equilibrium	Magnetic configuration, kinetic measurements
Separatrix to wall gap	Edge density profile
Growth/control of instabilities	Fluctuations over frequency range ~ 1 kHz - ~ 2 MHz
Burn onset and control	Neutron flux and beta
Plasma rotation profile	Toroidal rotation speed
Fuel species, impurity density	Density measurements of D, T, H and low-Z and high-Z impurities
Non-inductive current drive	Current density profile
Excess growth of helium in core	Helium density
Fuel and fuel ratio control	Exhaust neutral densities
Auxiliary power input performance	Many measurements to be determined
Density control at start-up, auxiliary heating permissives	Density (probably profile)

Table 2a: Area estimates for diagnostics proposed to provide measurements for device protection

Parameter to be controlled	**Contributing plasma measurement**	**Diagnostic**	**Space Required (m^2)**
Disruptive instabilities	Disruption precursors	Not known (two)	0.2
	Large ELMs	Mirnov Coils (Magnetics)	None
		D-alpha detector	0.084
Intense local heat load	High local first wall/divertor plate temperatures	Infra-red camera systems (6 locations)	0.096
Plasma beta	Magnetic	Diamagnetic Loop (Magnetics)	None
	From kinetic measurements		
	n_e profile	5-channel 10 μ radial interferometer	0.110
		30 Hz Lidar Thomson scattering	0.076
	T_e profile	Radial single-sightline ECE system	0.081
		30 Hz Lidar Thomson scattering	No additional
	I_p, plasma current	Rogowski Coil (Magnetics)	None
Start-up initial conditions	Pressure and partial pressures	Pressure gauges and Residual Gas Analyzers	No additional (in pumping ducts)
Start-up localization	Magnetic equilibria	Magnetic loops inside vacuum vessel	None
Profile control (auxiliary heating, fuelling)	n_e profile	5-channel 10 μ radial interferometer	No additional
		30 Hz Lidar Thomson scattering	No additional
	T_e profile	Radial single-sightline ECE system	No additional
		30 Hz Lidar Thomson scattering	No additional
	Poloidal rotation profile	X-ray crystal spectrometer	0.0.24
	Toroidal rotation profile	X-ray crystal spectrometer	0.24
Development of excess alpha-particle loss	Lost- alphas, high local wall temperature	Infra-red 1^{st}-wall temperature (two)	0.032
Significant loss of plasma (disruption, etc.)	Halo currents	Loops inside vessel	No additional
	High local wall temperatures	Infra-red 1^{st}-wall temperature	No additional
	Divertor strike-point temperatures	Infra-red divertor monitors (two)	0.032
	Runaway electrons	Not known	0.1

Table 2b: Area estimates for diagnostics proposed to provide plasma control

Parameter to be controlled	**Contributing plasma measurement**	**Diagnostic**	**Space Required (m^2)**
Plasma equilibrium	Magnetic configuration	Magnetic loops inside vacuum vessel	None
		Position reflectometer	0.081
	Kinetic measurements	See above for measurements	No additional
Separatrix to wall gap	Edge density profile	Position reflectometer	No additional
Growth/control of instabilities	Fluctuations over frequency range ~ 1 kHz - ~ 30kHz	Mirnov Coils (Magnetics)	None
	~30kHz - ~ 2MHz	Reflectometer	0.162
		Radial single-sightline ECE system	No additional
Burn onset and control	Plasma beta	See above for measurements	No additional
	Integrated neutron flux	Neutron fission chambers	0.08
	Neutron source profile	5-channel neutron camera	0.30
Current density profile	j-profile	Not known	0.1
Plasma rotation profile	Toroidal rotation speed	X-ray crystal spectrometer	No additional
Fuel and fuel ratio control	Exhaust neutral densities	Residual gas analyzers in pumping ducts	No additional
	Core n_T/n_D	Not known (fast-wave reflectometer?)	0.1
Excess growth of helium in core	Helium density in core	Not known	0.1
Impurity density	Low-Z impurities	UV spectroscopy (edge only)	0.11
	High-Z impurities	X-ray spectroscopy	0.24
Auxiliary power input performance	Visible and IR-camera views of launching antennae	Visible and IR cameras	Partial additional
	Measurements to be determined with choice of heating and current drive	Not known (two)	0.20
Density control at start-up, auxiliary heating permissives	Density (probably profile)	5-channel interferometer	0.11

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